OMAHA PUBLIC POWER DISTRICT

NUCLEAR ANALYSIS
RELOAD CORE ANALYSIS METHODOLOGY OVERVIEW

OPPD-NA-8301-NP
Rev. 04
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ABSTRACT

This document is a Topical Report describing Omaha Public Power District's reload core analysis methodology for application to the Fort Calhoun Station Unit No. 1.

The report provides an overview of the District's reload core methodology. Analyses performed by the District and its contractors are described. Details of the thermal hydraulic methodology which were previously submitted to the NRC are provided.

PROPRIETARY DATA CLAUSE

This document is the property of Omaha Public Power District (OPPD) and contains the nonproprietary information, indicated by brackets, developed by Advanced Nuclear Fuels Corp (ANF), ABB Combustion Engineering (CE) and Westinghouse Electric Corporation (W). The ANF, CE, and Westinghouse information was purchased by OPPD under proprietary information agreements.

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REVISION	DATE
00	September 1983
01	June 1985
02	November 1986
03	April 1988
04	April 1991

OMAHA PUBLIC POWER DISTRICT
RELOAD CORE METHODOLOGY OVERVIEW

1.0 INTRODUCTION

Analyses done to license reload cores for Fort Calhoun Station consist of the analysis performed by the Omaha Public Power District and the analysis performed by the nuclear fuel vendor. The current nuclear fuel vendor is Westinghouse Electric Corporation (W); however, future reload fuel may be potentially supplied by any of the four U.S. PWR nuclear fuel vendors: Advanced Nuclear Fuels Corp. (ANF), ABB Combustion Engineering (CE), Westinghouse, or Babcock and Wilcox. The following sections discuss the reload analyses and consolidate information about the District's methodology previously submitted.

2.0 FUEL SYSTEM DESIGN

The fuel assembly mechanical design and analysis are performed by the nuclear fuel vendor. The fuel mechanical design and design methods utilized for Fort Calhoun Station by Westinghouse are described in Reference 2-1. Combustion Engineering, the co-resident fuel in the mixed core, fuel mechanical design and design methods are discussed in References 2-2 and 2-3.

In an effort to further reduce the neutron flux to the reactor vessel welds, full length Hafnium flux suppression rods, which are similar to the part length poison rods utilized in Cycle 1[illegible], will be incorporated into the fuel loading pattern. The poison rods are composed of hafnium metal over the full length of the active fuel. Inert material comprises the balance of the rod. They will reside in the outer guide tubes of quarter core assembly numbers 1, 2 and 8. The fuel system design will also incorporate four natural uranium fuel assemblies in quarter core location number 14 for additional neutron flux reduction to the critical welds.

3.0 NUCLEAR DESIGN

The District's nuclear design methodology is discussed in Reference 3-1.

3.1 Fuel Management

The reload core fuel management is performed by the District. Current fuel management schemes are selected to reduce flux to the reactor pressure vessel welds.

3.0 NUCLEAR DESIGN (Continued)

3.2 Power Distribution Measurement

The District utilizes the CE methodology (Reference 3-2) to measure the power distributions. This methodology is discussed in the Cycles 5 and 6 reload submittals and approved in the SER's for these fuel cycles (References 3-3 and 3-4).

3.3 Uncertainties and Allowances

The power distribution uncertainties which are included in the overall analysis of reload cores are:

Parameter	Uncertainty
3D Peak, Fq 3-D	6.2%
Integrated Radial Peak, F_r	6.0%
Planar Radial Peak, F_{xy}	5.3%

These values are approved for use in CENPD-153-P (Reference 3-5). A more detailed discussion of the treatment of uncertainties and allowances can be found in Reference 3-6.

3.4 Physics Safety Related Data

The physics safety related data are produced using the methodology discussed in Reference 3-1.

4.0 THERMAL HYDRAULIC DESIGN

4.1 Steady State DNBR Analysis

The steady state DNBR analysis is performed by the District using the TORC/CETOP/CE-1 methodology (References 4-1, 4-2, 4-3, 4-4 and 4-5). This methodology was approved for use by the District in Reference 4-6.

4.1.1 Grid Spacer Loss Coefficients

The analysis utilizes a D-TORC model with explicit representation of the loss coefficients associated with CE and W fuel assemblies. The nominal grid loss coefficients used in thermal hydraulic analysis are:

4.0 THERMAL HYDRAULIC DESIGN (Continued)

4.1 Steady State DNBR Analysis (Continued)

4.1.1 Grid Spacer Loss Coefficients (Continued)

W Spacer	CE Spacer	
[	]	Loss Coefficient (K)

Re = Reynolds Number

These values were obtained by Westinghouse using single phase pressure drop testing of a Westinghouse test assembly and a typical CE 14 x 14 assembly. Single phase hydraulic loss coefficients, previously transmitted in Reference 4-7, contained a [ ] while the values given above are best estimate values. Because of the sensitivity of DNBR calculations to the difference in spacer grid loss coefficients, the District utilizes the Reynolds number expression for loss coefficients. This provides the most accurate representation of the pressure drop across each spacer grid in the assembly. Thus, the cross flows between adjacent assemblies in the region of the spacer grid are accurately modeled.

The spacer grid geometries for the CE and W spacer grids are shown in References 2-2 and 2-1 , respectively. The spacer grid envelope for both the CE and W grids is 8.115 inches by 8.115 inches. The axial location of the CE, W and ANF spacer grids is shown in Figure 4-3.

In D-TORC calculations, the spacer grid loss coefficient for a channel corresponds to the assembly type whenever a channel represents a single assembly or a portion of an assembly. The choice of loss coefficient for lumped channels in D-TORC is made such that the minimum flow is provided to the limiting fuel assembly. The CETOP model employs the spacer grid loss coefficient for the limiting assembly calculated in D-TORC. The inlet flow fraction of the CETOP model is tuned such that the CETOP model produces conservative results with respect to the D-TORC model, which models all fuel assemblies.

4.1 Steady State DNBR Analysis (Continued)

4.1.2 CE-1 Correlation

The District utilizes the CE-1 correlation for DNBR calculations. The range of data in the data base for the CE-1 correlation is contained in References 4-1 and 4-2. The range of parameters for the CE-1 correlation and corresponding ranges for the CE and W assemblies are shown in Table 4-1. Because the data for the W fuel assembly is within the range specified in the CE-1 data base, the use of CE-1 correlation is appropriate for the W fuel.

4.1.3 D-TORC and CETOP Models

The District utilizes the D-TORC code (Reference 4-3) and the CETOP code (Reference 4-4) to perform thermal hydraulic analysis for the Fort Calhoun reload core. The fraction of inlet flow to the hot assembly in the CETOP model is adjusted such that the model yields appropriate MDNBR results when compared with results of D-TORC analysis for a given range of operating conditions. The fraction of inlet flow is determined for each reload core. The use of this methodology was approved for use by the District in Reference 4-6.

The following paragraphs discuss the application of the CETOP code to the Fort Calhoun reactor. Examples are for the Cycle 8 core (which contained ANF and CE fuel types).

Thermal margin analysis utilizing the CETOP model is supported by comparing its predictions for Fort Calhoun Station with those obtained from a detailed TORC analysis. Several operating conditions were arbitrarily selected for this demonstration; they are representative, but not the complete set, of conditions which would be considered for a normal DNB analysis.

4.0 THERMAL HYDRAULIC DESIGN (Continued)

4.1 Steady State DNBR Analysis (Continued)

4.1.3 D-TORC and CETOP Models (Continued)

A thermal margin model for 1500 MWt for Fort Calhoun Unit No. 1 was developed for the following operating ranges.

Inlet Temperature	45 to 600 °F
System Pressure	1750 to 2400 psia
Primary System 4-Pump Flow Rate,	
(LCO = 196,000 gpm)	80 to 120%
Axial Power Distribution	-0.517 to +.526 ASI

The detailed thermal margin analyses were performed for the sample core using the radial power distribution and detailed TORC model shown in Figures 4-4, 4-5, and 4-6. The appropriate spacer grid loss coefficient was applied to each "assembly" channel or partial assembly channel in each stage. In stage 1, lumped channel 28 utilized the CE spacer grid loss coefficient because the channel was predominantly composed of CE fuel. Lumped channel 26 and 27 utilized the ANF spacer grid loss coefficient because either the channel was composed of entirely ANF fuel or contained a single CE assembly not on a boundary between channels. The axial power distributions are given in Figure 4-7. These distributions were the most limiting ones generated for the length of the cycle and for the various power dependent insertion limits examined. The core inlet flow and exit pressure distributions used in the analysis were based on the flow model test results given in Figures 4-8 and 4-9. The results of the detailed TORC analyses are given in Table 4-2. The Cycle 14 TORC and CETOP models incorporate the appropriate changes for W fuel design parameters. The same methods apply to the mixed CE and W core that were applied to the CE and ANF mixed core.

The CETOP design model was a total of four thermal hydraulic channels to model the open-core fluid phenomena. Figure 4-10 shows the layout of these channels. Channel 2 is a quadrant of the hottest assembly which represents

4.0 THERMAL HYDRAULIC DESIGN (Continued)

4.1 Steady State DNBR Analysis (Continued)

4.1.3 D-TORC and CETOP Models (Continued)

the average coolant conditions for the remaining portion of the core. The boundary between channels 1 and 2 is open for crossflow; the remaining outer boundaries of channel 2 are assumed to be impermeable and adiabatic. Channel 2 includes channels 3 and 4. Channel 3 lumps the subchannels adjacent to the MDNBR hot channel 4. The "hot" assembly determined from D-TORC analysis was an ANF assembly. Since CETOP models a quadrant of the "hot" assembly, the ANF spacer grid loss coefficient was used in the analysis.

The CETOP model described above was applied to the same cases as the detailed TORC analyses. The results from the CETOP model analyses are compared with those from the detailed analyses in Table 4-2. It was found that a constant inlet flow split providing hot assembly inlet mass velocity of [ ] of the core average value is appropriate for 4-pump operation so that MDNBR results predicted by the CETOP model are either conservative or accurate for the Cycle 8 core. The uncertainties associated with the thermal hydraulic analysis are combined statistically (Reference 4-8). In this method, the impact of component uncertainties on DNBR is assessed and the SAFDL is increased to include the effects of the uncertainties.

5.0 POSTULATED ACCIDENTS AND TRANSIENTS

The postulated accidents and transients are analyzed using the methodology discussed in Reference 5-1.

6.0 SETPOINT GENERATION

The District utilizes the methodology discussed in CENPD-199-P (Reference 6-1) to generate setpoints for Fort Calhoun Station. The District's reactor physics methodology is discussed in Reference 6-2.

6.0 SETPOINT GENERATION (Continued)

The scram reactivity curves are produced using the QUIX code. The power-to-fuel design limit on centerline melt is derived using the QUIX code with the appropriate combinations of planar radial peaking factor, F_{xy}^T, and axial power distribution.

The thermal margin analysis is done using the CETOP code with the appropriate combinations of the integrated radial peaking factor, F_R^T, axial power distribution, RCS inlet temperature, and RCS pressure.

The Fort Calhoun RPS utilizes the "standard" local power density trip and TM/LP trip. The sequential CEA withdrawal is analyzed using the methods described in Reference 6-1 and not included in the TM/LP trip considerations. The RCS depressurization event provides the transient analysis input into the TM/LP trip.

6.1 Incore LCO Monitoring

The Better Axial Shape Selection System (BASSS) monitors the Limiting Conditions for Operation on peak linear heat rate and departure from nucleate boiling using as input the data available from the MINI-CECOR code and the plant computer. This arrangement is similar to the one used by Baltimore Gas and Electric at their Calvert Cliffs Units, and described in the Combustion Engineering Setpoint Topical (Ref. 6-1).

MINI-CECOR is a plant computer version of Combustion Engineering's CECOR code. It is used in this application to synthesize the following parameters from readings of the fixed in-core detectors:

1. The three-dimensional power peaking factor (F_q)
2. The core average axial shape index ($\bar{I}$)
3. The total planar radial peaking factor (F_{XY}^T)
4. The total integrated radial peaking factor (F_R^T)

These inputs to BASSS are descriptive of the existing core power distribution.

The inputs to BASSS obtained from the plant computer are the following:

1. Measured core power level
2. Percent insertion of the lead CEA regulating group.

6.0 SETPOINT GENERATION (Continued)

6.1 Incore LCO Monitoring (Continued)

BASSS consists of two algorithms: one for peak linear heat rate monitoring and another for DNB monitoring. The peak linear heat rate algorithm uses the 3-D power peaking factor and the measured core power level to calculate the core peak linear heat rate. The algorithm applies appropriate uncertainties and allowances (per the Technical Specifications) to the 3-D peaking factor. The measured peak linear heat rate is compared to the monitoring limit, which is based on both LOCA and AOO transient analysis considerations, and an alarm is activated when the monitoring limit is exceeded. The power operating limit on linear heat rate is also calculated and displayed as an indication of the available operating margin. The DNB algorithm is an improvement over the excore ASI monitoring system in that it uses in-core axial shape index, CEA group position and the radial peaking factors to establish the plant's power operating limit. An alarm is activated when the power operating limit is exceeded. A gain in operating margin results from the following.

1. A reduction in ASI uncertainty due to the use of in-core ASI versus ex-core ASI.

2. Knowledge of the actual CEA group position versus the ex-core system's assumption that the CEAs are inserted to the PDIL's transient insertion limit.

3. Knowledge of the actual radial peaking factors versus the ex-core system's assumptions that radial peaks are at the Technical Specification limits.

6.0 SETPOINT GENERATION (Continued)

6.1 Incore LCO Monitoring (Continued)

BASSS is also provided with the capability to monitor the Limiting Condition for Operation on F_{xy}^T and F_r^T. If the Technical Specification for Operation on F_{xy}^T or F_r^T are exceeded during normal plant operation, BASSS will activate an alarm and calculate the proper trade-off with maximum allowed power that ensures that the Axial Power Distribution and Thermal Margin/Low Pressure Trips remain conservative. An alarm is activated if the measured power level is higher than the allowed power level.

6.2 Uncertainties

The uncertainties are treated statistically in the District's setpoint analysis as identified in references 6-1 and 6-3.

7.0 CORE OPERATING LIMITS REPORT

The core operating limits report (COLR) will be incorporated into the plant Technical Specifications as part of the Cycle 14 reload license application. The COLR will utilize the guidance provided in Reference 7-1. The reload analysis for Fort Calhoun to generate the COLR values will utilize the methods described in References 7-2, 7-3 and 7-4. The use of these NRC approved methodologies does not permit substantial discretion on the part of OPPD and does not require substantial engineering judgement to be utilized to derive the cycle specific parameter limits included in the COLR.

7.0 CORE OPERATING LIMITS REPORT (Continued)

The COLR will consist of the following items:

- Thermal Margin/ Low Pressure LSSS for 4 Pump Operation
- Refueling Boron Concentration
- Limiting Conditions for Operation for Excore Monitoring of LHR
- Power Dependent Insertion Limit (PDIL)
- Unrodded Integrated Radial Peaking Factor (F_r)
- Unrodded Planar Radial Peaking Factor (F_{xy})
- Allowable Peak Linear Heat Rate vs. Burnup
- F_{xy}^T, F_r^T and Core Power Limitations

In accordance with Reference 7-1 requirements, updates to the COLR during the operating cycle will be issued to the NRC (NRR, Region IV and Senior Resident Inspector) concurrent with internal OPPD distribution.

8.0 REFERENCES

Section 2 References

2-1 "Westinghouse Reload Fuel Mechanical Design Evaluation for the Fort Calhoun Station Unit 1", March 1991.

2-2 "Omaha Batch M Reload Fuel Design Report", CEN-347(0)-P, Rev. 01, January 1987.

2-3 "Fuel Rod Maximum Allowable Gas Pressure", CEN-372-P-A, March 1990.

Section 3 References

3-1 "Reload Core Analysis Methodology, Neutronics Design Methods and Verification", OPPD-NA-8302-P-A, Rev. 02, April 1988.

3-2 "INCA, Method of Analyzing Incore Detector Data in Pressurized Water Reactors", CENPD-145-P, April 1, 1975.

3-3 Letter from R.W. Reid (NRC) to T.E.Short (OPPD), December 5, 1978.

3-4 Letter from R.W. Reid (NRC) to W.C. Jones (OPPD), April 1, 1980.

3-5 "INCA/CECOR Power Peak Uncertainty", CENPD-153-P, Revision 1-P-A, May 1980.

3-6 "Statistical Combination of Uncertainties", CEN-257(0)-P-A, Parts 1 and 3, November 1983.

Section 4 References

4-1 "CE Critical Heat Flux", CENPD-162-P-A, Part 1, Combustion Engineering, September 1976.

4-2 "CE Critical Heat Flux", Part 2, CENPD-207-P-A, Combustion Engineering, June 1976.

4-3 "A Computer Code for Determining the Thermal Margin of a Reactor Core," CENPD-161-P, TORC Code, Combustion Engineering, July 1975.

4-4 "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 & 2," CEN-191(B)-P, Combustion Engineering, December 1981.

4-5 Letter from Cecil Thomas (NRC) to Mr. A. E. Scherer (CE), November 2, 1984

8.0 REFERENCES (Continued)

Section 4 References (Continued)

4-6 Letter from R. A. Clark (NRC) to W. C. Jones (OPPD), March 15, 1983.

4-7 Letter from W. C. Jones (OPPD) to R. W. Reid (NRC), December 4, 1979.

4-8 "Statistical Combination of Uncertainties," CEN-257(0)-P-A, Part 2, Novemt er 1983.

Section 5 References

5-1 "Reload Core Analysis Methodology, Transient and Accident Analysis Methc is and Verification," OPPD-NA-8303-P, Rev. 03, April 1991.

Section 6 References

6-1 "CE Setpoint Methodology," CENPD-199-P, Revision 1-P-A, January 1986.

6-2 "Reload Core Analysis Methodology, Neutronics Design Methods and Verific ition," OPPD-NA-8302-P-A, Rev. 02, April 1988.

6-3 "Statistical Combination of Uncertainties," CEN-257(0)-P-A, Parts 1, 2, and November 1983.

Section 7 References

7-1 "Removal of Cycle-Specific Parameter Limits from Technical Specifications" NRC Generic Letter 88-16, October 4, 1988.

7-2 "Reload Core Analysis Methodology Overview," OPPD-NA-8301-P, Rev. 04 April 1991.

7-3 "Reload Core Analysis Methodology, Neutronics Design Methods and Verific ition," OPPD-NA-8302-P-A, Rev. 02, April 1988.

7-4 "Reload Core Analysis Methodology, Transient and Accident Analysis Methc is and Verification," OPPD-NA-8303-P, Rev. 03, April 1991.

TABLE 4-1

PARAMETER RANGES OF THE SOURCE DATA FOR THE CE-1 CHF CORRELATION AND THE RANGE OF WESTINGHOUSE, CE AND ANF 14 x 14 FOR FORT CALHOUN VALUES

PARAMETER	CORRELATION RANGE	CE RANGE	ANF RANGE	WESTINGHOUSE RANGE
Pressure (psia)	1785 to 2415	N/A	N/A	N/A
Local Coolant Quality	-.16 to .20	N/A	N/A	N/A
Local Mass Velocity ($lb_m/hr-ft^2$)	0.87×10^6 to 3.21×10^6	N/A	N/A	N/A
Subchannel Wetted Equiv. Diameter (in)	.3588 to .5447	.4043 to .5449	.4010 to .5402	.40[illegible]3 to .5[illegible]49
Subchannel Heated Equiv. Diameter (in)	.4713 to .7837	.5334 to .7840	.5270 to .7760	.53[illegible]4 to .7[illegible]40
Heated Length (in)	84 to 150	128	128	1[illegible]8
Grid Spacing (in)	14.2 to 18.25	16.8	16.8	1[illegible]8

TABLE 4-2

COMPARISONS BETWEEN TORC AND CETOP-D

Operating Parameters					MDNBR		Quality at MDNBR		A[illegible] Elev. of M[illegible]NBR (in)	
Pres (psia)	Inlet Temp (F)	Avg Mass Velocity (1)	Core Avg. Heat Flux (2)	Shape Index (ASI)	Detailed TORC Relative Flow in Loc 5	CETOP Inlet Flow Factor [ ]	Detailed TORC Relative Flow in Loc 5	CETOP Inlet Flow Factor [ ]	Detaile TORC	CETOP
1750	450	1.7432	242409	-.517	[					]
2100	450	1.7432	257008	-.517						
2250	450	1.7432	261195	-.517						
2400	450	1.7432	264118	-.517						
2100	545	2.1790	216494	-.517						
1750	600	1.7432	149283	-.206						
2100	600	1.7432	168727	-.206						
2250	600	1.7432	176398	-.206						
2400	600	1.7432	184260	-.206						
2100	545	2.1790	257118	-.206						
2100	545	2.1790	282778	.004						
2100	545	2.1790	298644	.203						
1750	450	1.7432	295262	.527						
1750	545	1.7432	227063	.527						
1750	600	1.7432	147319	.527						
2100	545	2.1790	255014	.527						

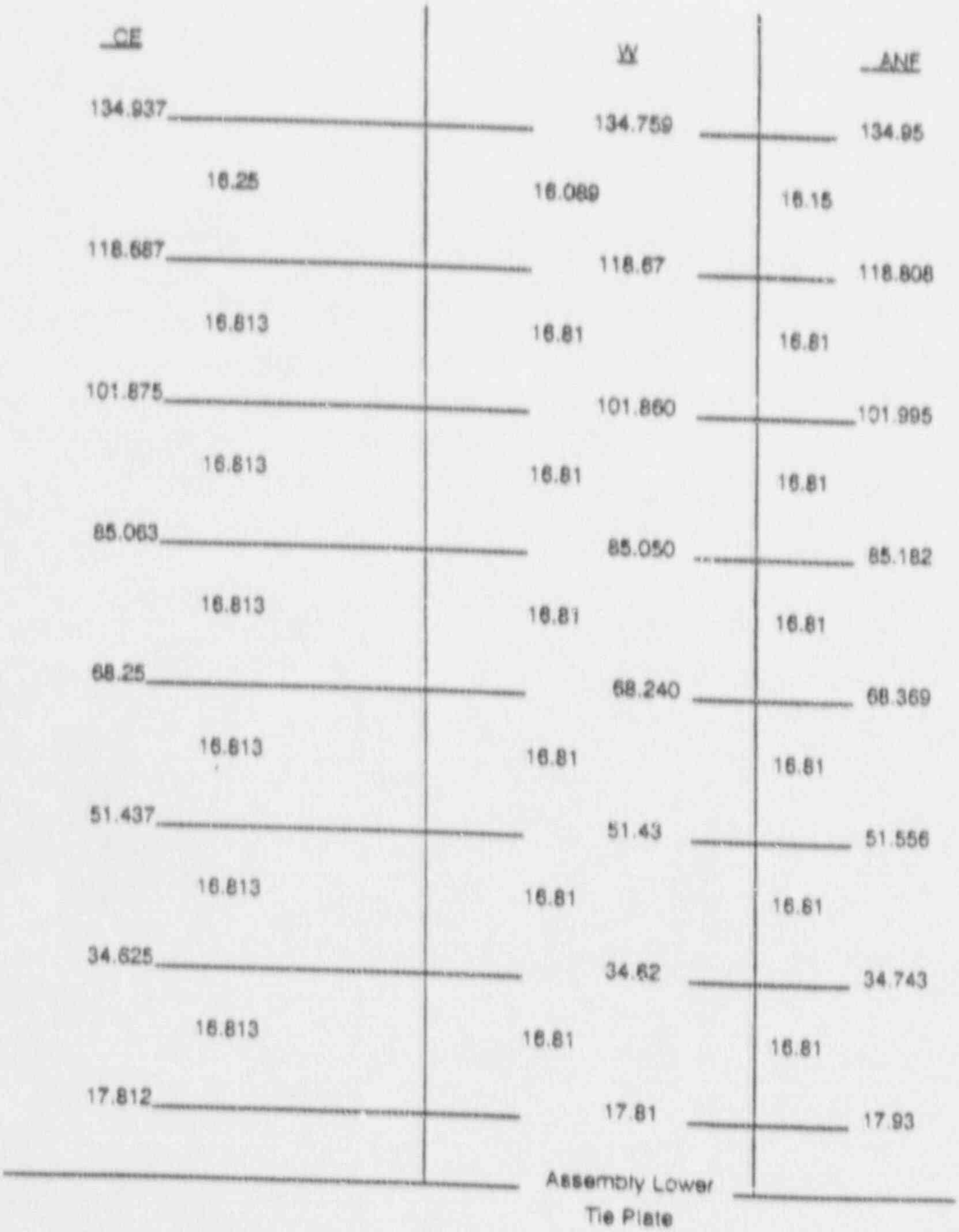
(1) (10^6 lb_m/hr-ft^2)
(2) (BTU hr-ft^2)

DELETED

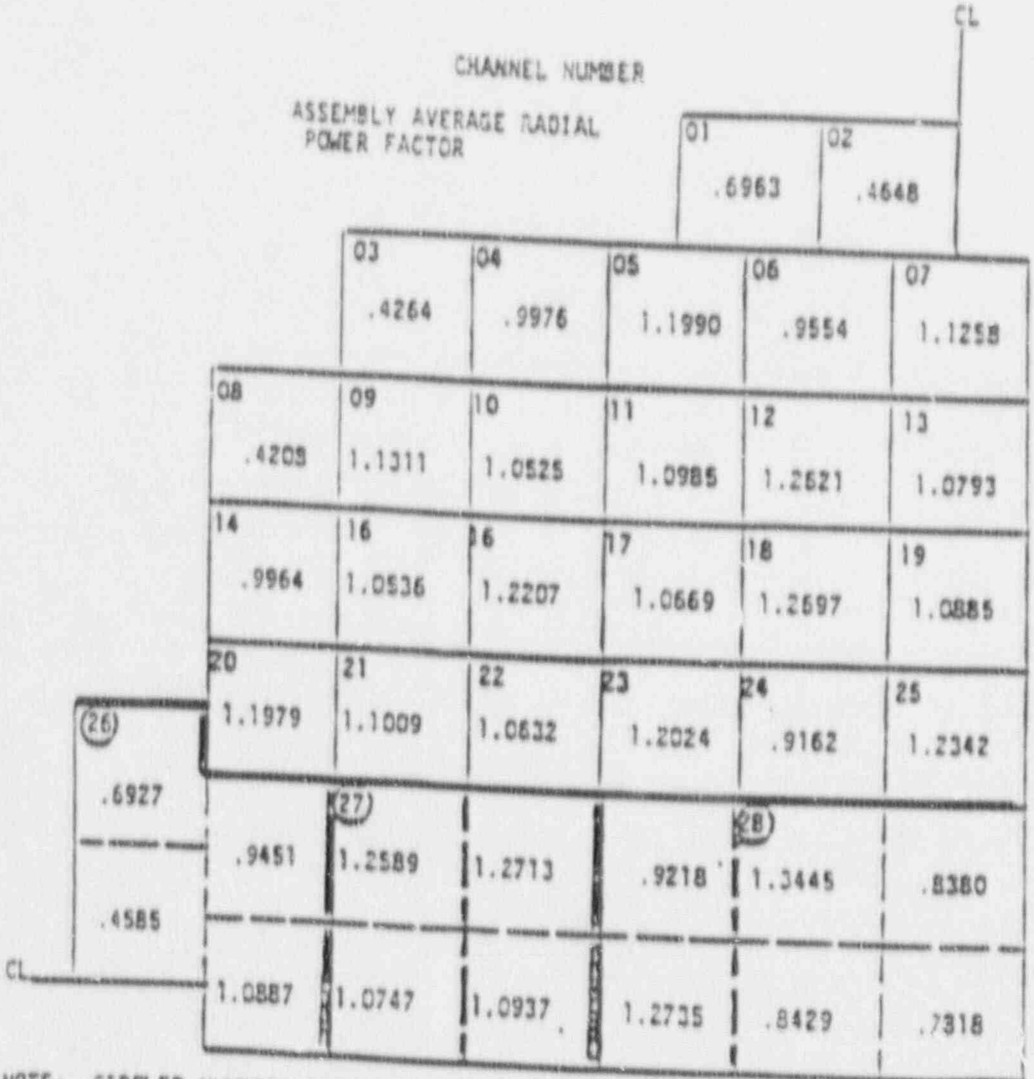
CE FUEL SPACER GRID	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 4-1

DELETED

ANF FUEL SPACER GRID	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 4-2

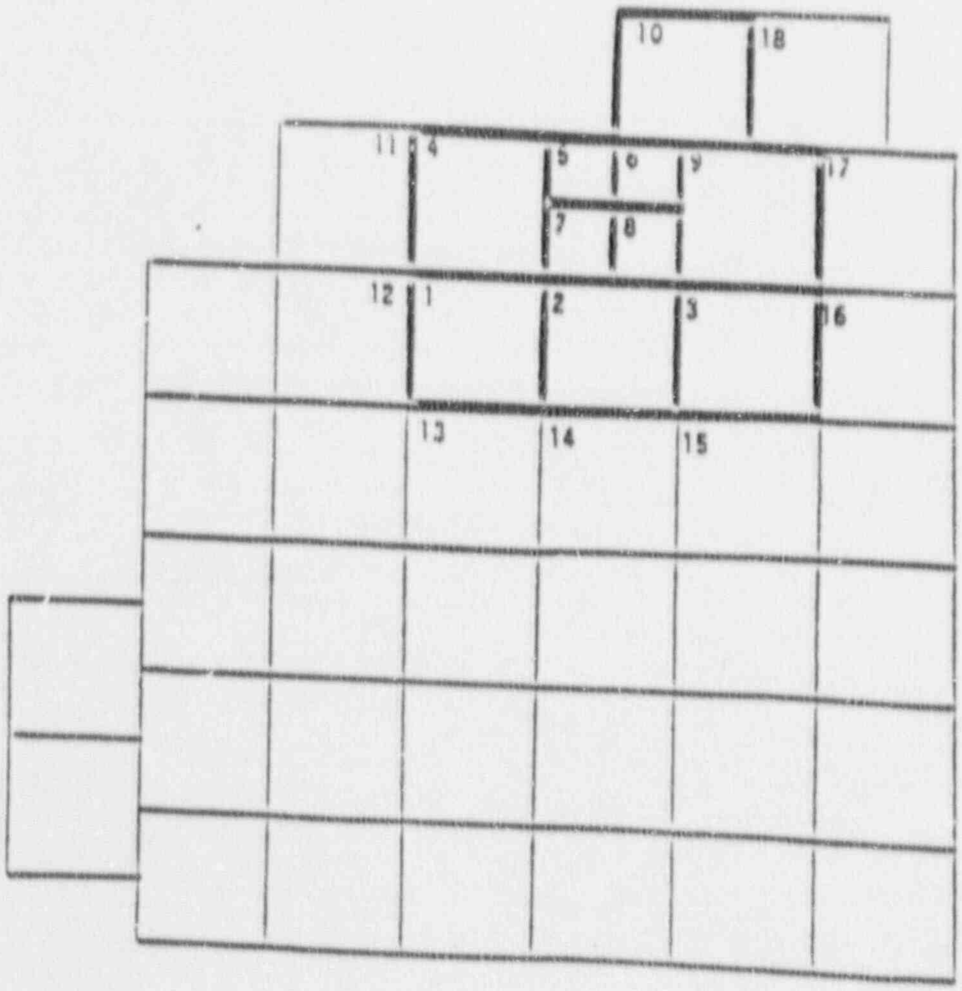


AXIAL LOCATION OF FUEL ASSEMBLY SPACER GRIDS	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 4-3

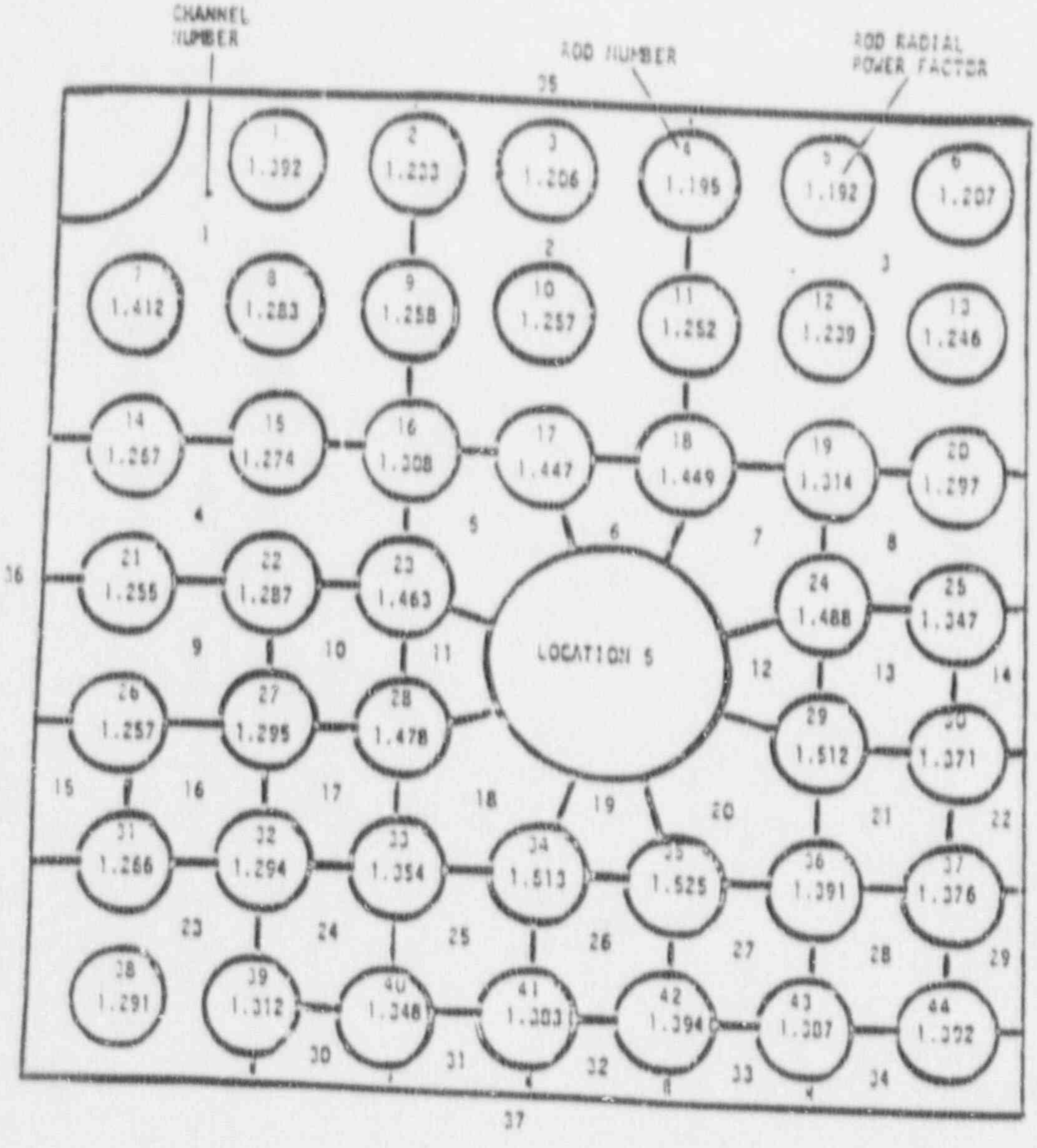


NOTE: CIRCLED NUMBERS DENOTE "LUMPED" CHANNEL

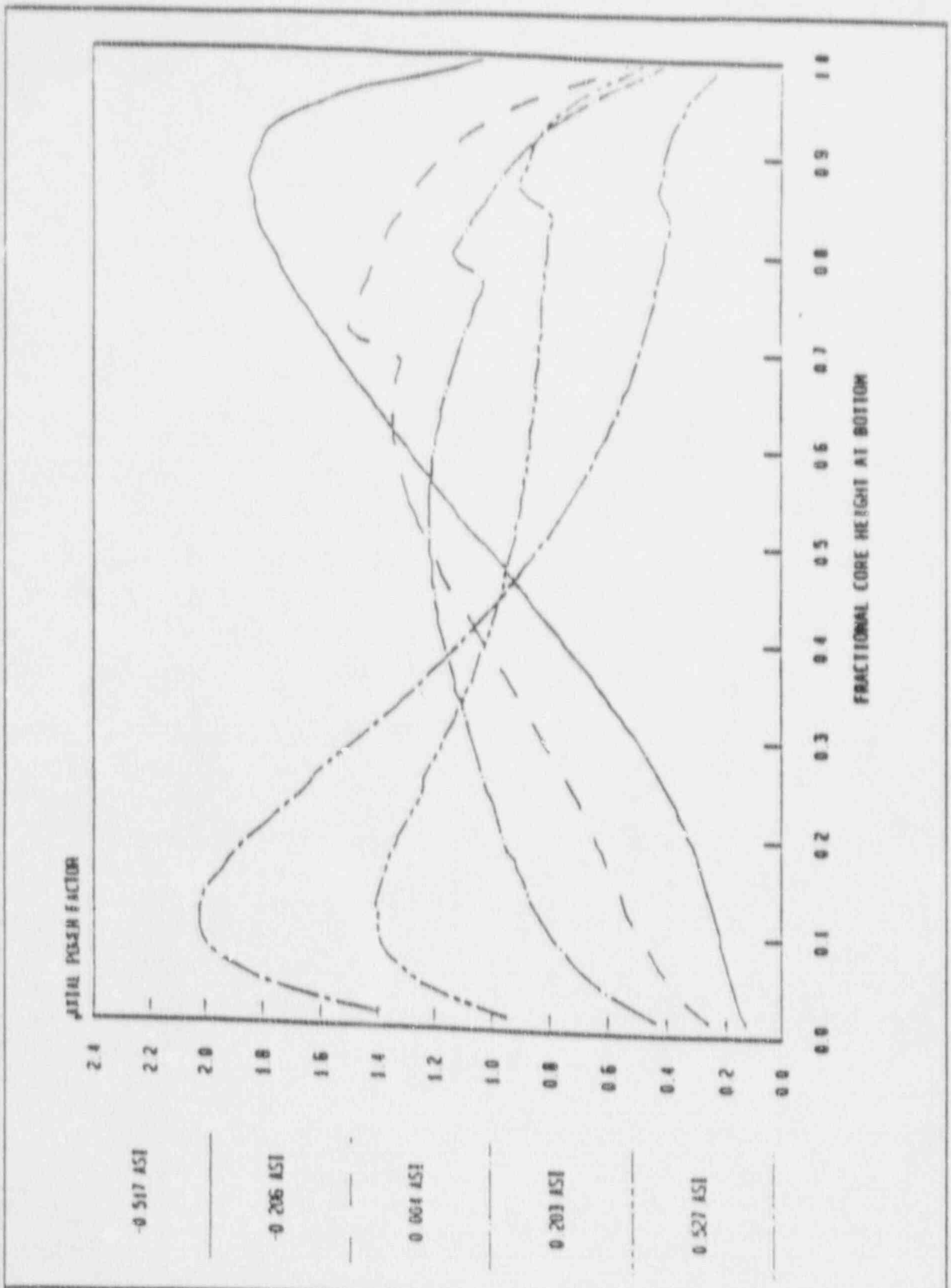
STAGE 1 TORC CHANNEL GEOMETRY FOR FCS UNIT NO.1	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 4-4



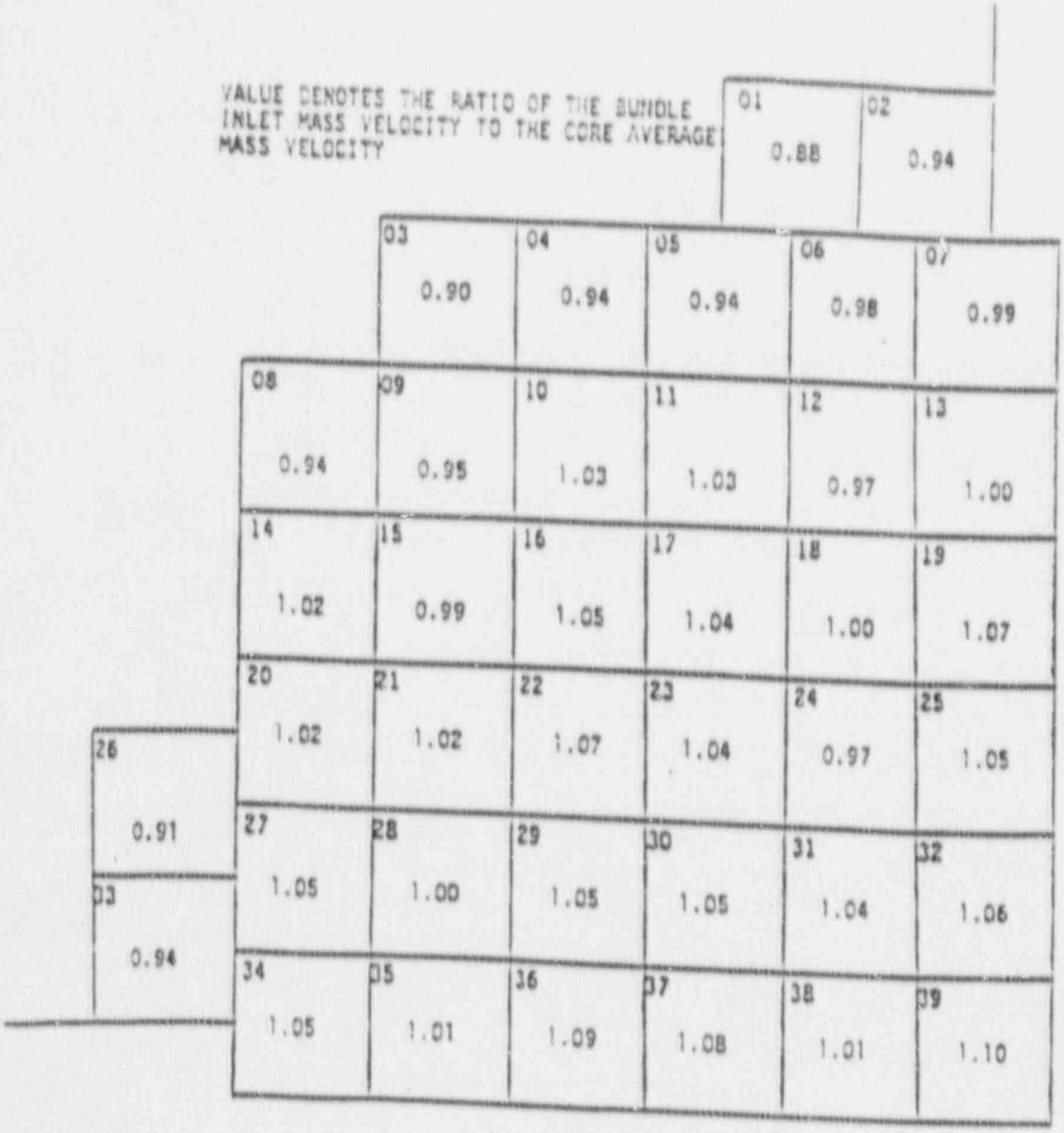
STAGE 2 TORC CHANNEL GEOMETRY FOR FCS UNIT NO.1	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 4-5



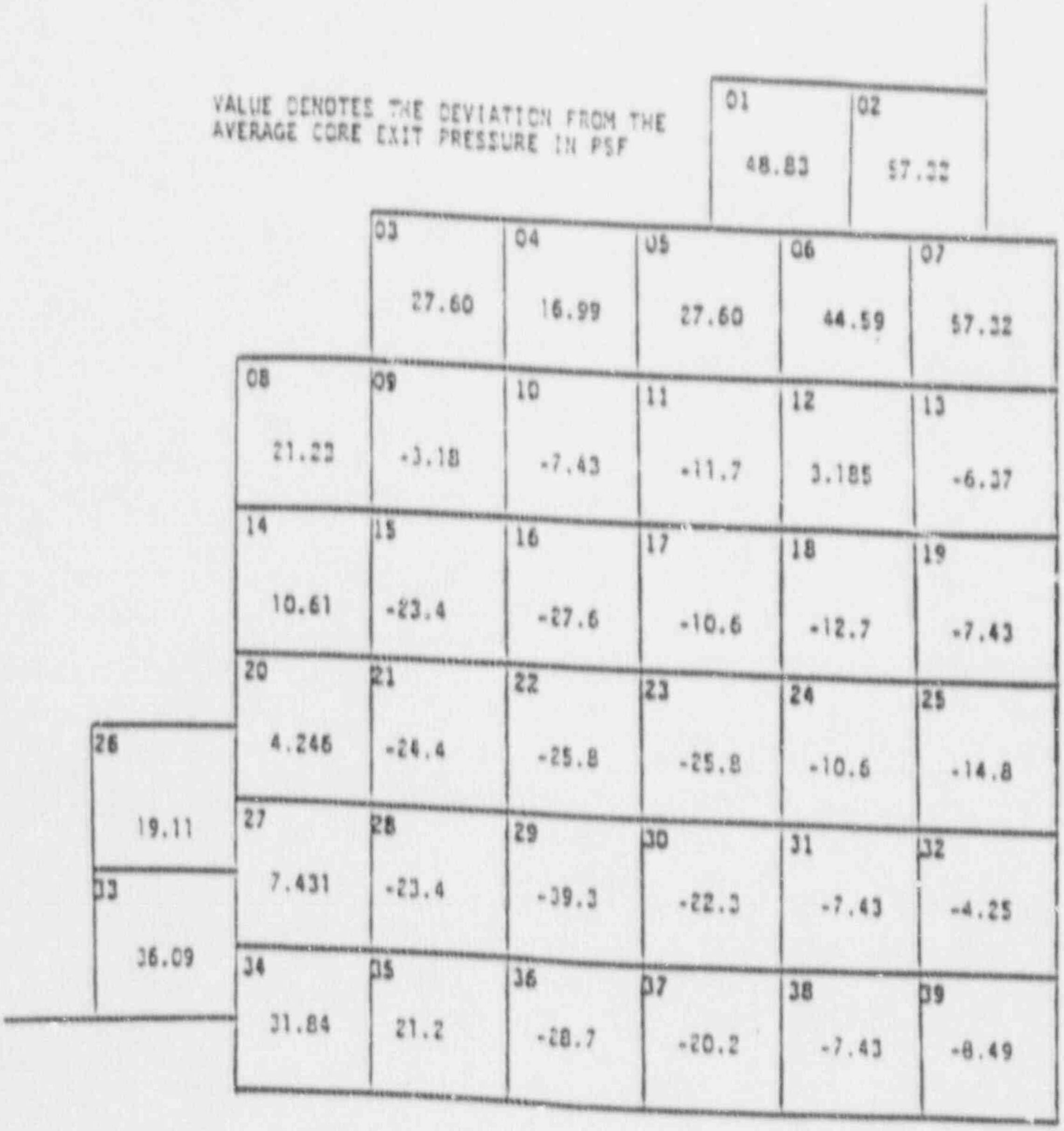
STAGE 3 TORC CHANNEL GEOMETRY FOR FCS UNIT NO.1	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 4-6



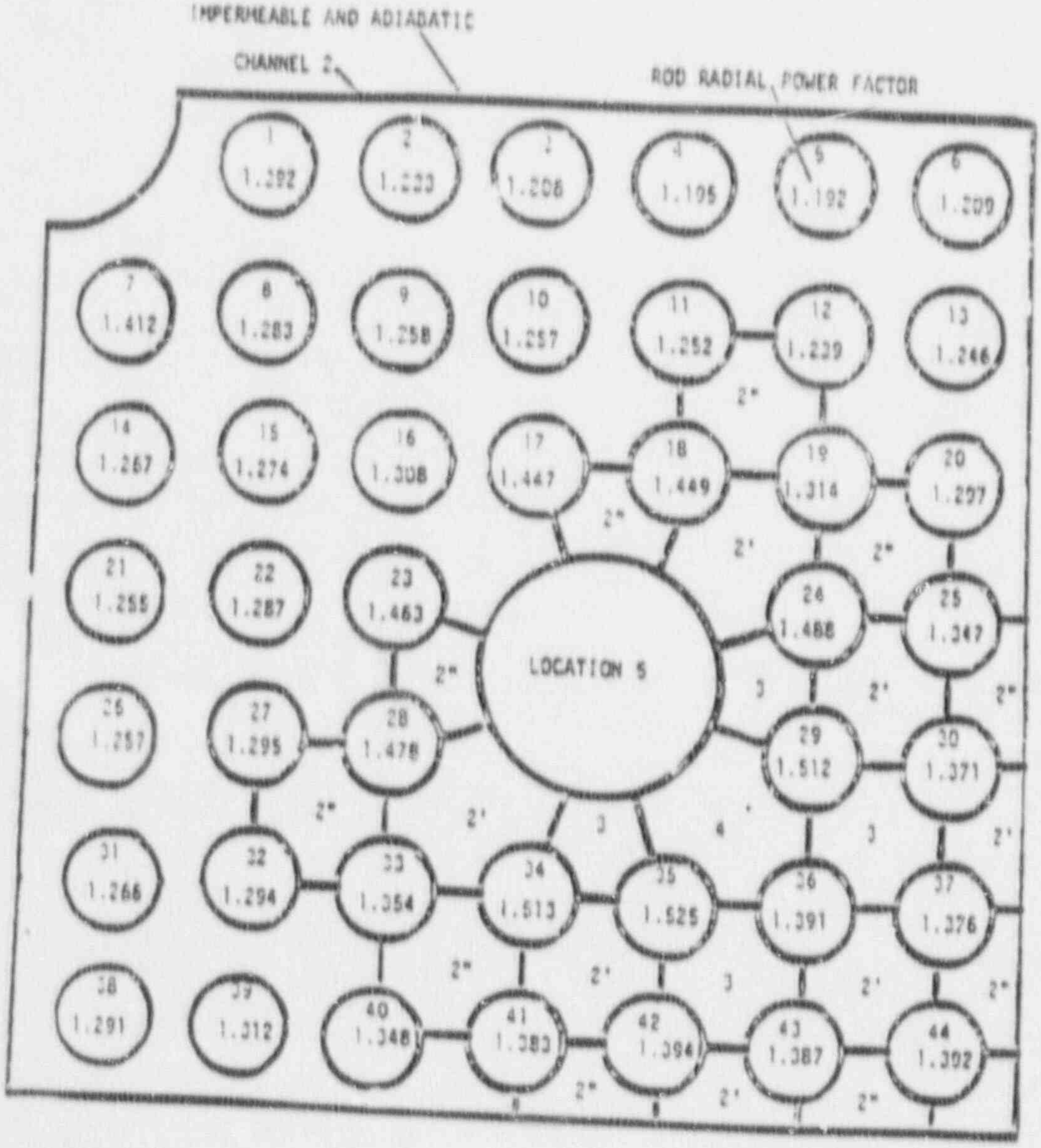
AXIAL POWER DISTRIBUTIONS FORT CALHOUN CYCLE 8	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 4-7



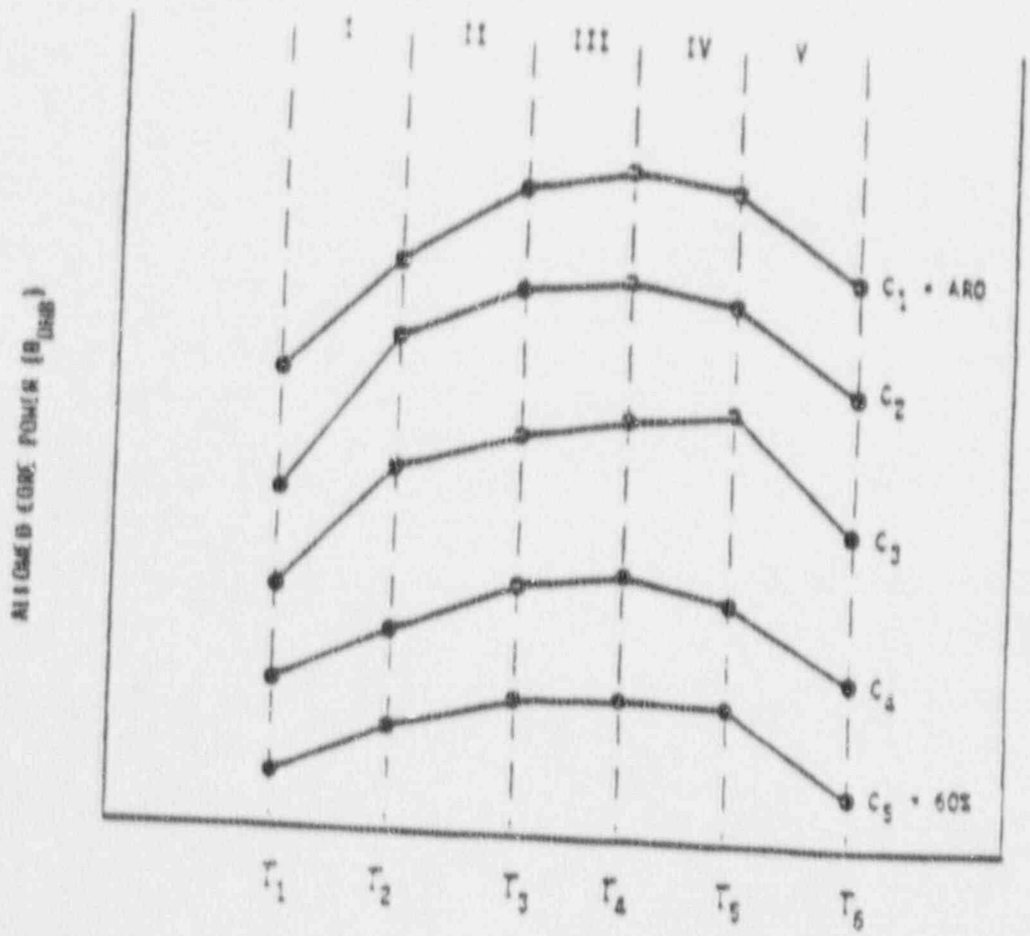
INLET FLOW DISTRIBUTION FOR FCS, 4-PUMP OPERATION	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 4-8



EXIT PRESSURE DISTRIBUTION FOR FCS, 4-PUMP OPERATION	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 4-9



CETOP-D CHANNEL GEOMETRY (CHANNEL 1 NOT SHOWN)	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 4-10



ALGORITHM FOR DNB LCO MONITORING	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 6-1

Table 1

RELOAD CORE ANALYSIS METHODOLOGY OVERVIEW

OPPD-NA-8301-NP

REV. 04

Title Page	Changed the revision number and date.
All Pages	Updated revision number.
All Pages	Change all references of Combustion Engineering to ABB Combustion Engineering.
All Pages	Replaced references to ANF with references to Westinghouse (also denoted by W) where appropriate.
ii	Added Westinghouse Electric Corporation to Proprietary Data Clause.
iii	Updated Table of Contents to include Chapter 7 on Core Operating Limits Report
iv	Added reference to W (Westinghouse) to Table 4-1 on List of Tables
v	Deleted Figures 4-1 and 4-2 from list .
vi	Updated Revision Sheet.
1	Added reference to Westinghouse Electric Corporation as current fuel vendor. Added words "potentially" and "U.S." Changed paragraph 3 to discuss current PTS flux reduction efforts.
3	Changed reference to Figures 4-1 and 4-2 to References 2-2 and 2-1, respectively. Replaced ANF with W grid spacer loss coefficients and used test data for CE fuel.
4	Added historical detail on core loading during Cycle 8.
5	Added sentence regarding use of TORC with Westinghouse fuel assemblies.
7	Deleted "and excess load". Changed "events provide" to "event provides".
9, 10	Added Section 7.0 for Core Operating Limits Report.

Table 1 (Continued)

RELOAD CORE ANALYSIS METHODOLOGY OVERVIEW

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REV 04

11	Added references 2-1 and 2-3 to Section 2 references. Deleted references for ANF fuel from Section 2 reference list. Renumbered references. Added -A to NRC approved topical reports 3-1 and 3-6.
12	Updated reference 6-1 to the latest revision. Added references for Section 7. Added -A to NRC approved topical reports for reference 6-3.
13	Added parameter ranges for Westinghouse fuel.
17	Added dimensions for Westinghouse fuel grids, normalized distance to center of the grid.